Security Classification

DOCUMENT CONTROL DATA - R & D

(Security classification of title, body of abstract and indexing annotation must be entered when the overall report is classified)

1. ORIGINATING ACTIVITY (*Corporate author*) Naval Research Laboratory Washington, D.C. 20390	2a. REPORT SECURITY CLASSIFICATION Unclassified 2b. GROUP

3. REPORT TITLE

Procedures for Interpreting the Structural Implications of Radiation-Damage Surveillance Results on Nuclear Pressure Vessels

4. DESCRIPTIVE NOTES (*Type of report and inclusive dates*)

A final report on one phase of the problem; work continues.

5. AUTHOR(S) (*First name, middle initial, last name*)

L.E. Steele and C.Z. Serpan, Jr.

6. REPORT DATE	7a. TOTAL NO. OF PAGES	7b. NO. OF REFS
December 30, 1971	22	22

8a. CONTRACT OR GRANT NO. NRL Problem M01-14 b. PROJECT NO. RR022-11-41-5409 c. AEC-AT(49-5)-2110 d.	9a. ORIGINATOR'S REPORT NUMBER(S) NRL Report 7358 9b. OTHER REPORT NO(S) (*Any other numbers that may be assigned this report*)

10. DISTRIBUTION STATEMENT

Approved for public release; distribution unlimited.

11. SUPPLEMENTARY NOTES	12. SPONSORING MILITARY ACTIVITY Dept. of the Navy (Office of Naval Research) Arlington, Va. 22217, and U.S. Atomic Energy Commission, Washington, D.C. 20545

13. ABSTRACT

The structural implications of radiation effects to nuclear reactor pressure vessels are assessed primarily through surveillance programs in which the properties of the vessel are projected from an evaluation of small specimens of the vessel steel. In the USA, the current fracture-safe criterion requires that the vessel operating temperature, at certain stress levels, be at the FTE (Fracture Transition Elastic) temperature, defined as NDT+60°F(33°C), derived from surveillance measurements. Review of available data from five reactor surveillance programs indicates that this criterion is adequate for the vessels concerned. Complete assurance of fracture-safe operating conditions can be attained through a limit-analysis procedure that considers and integrates the effects of five factors: (a) the radiation-induced shift in transition temperature, (b) the initial shelf energy, (c) the radiation-reduced ductile shelf energy, (d) the effects of the fluence (and toughness) gradient through a thick vessel wall, and (e) the effects of thickness-induced mechanical constraint.

DD FORM 1 NOV 65 1473 (PAGE 1)

S/N 0101-807-6801

Security Classification

Security Classification

14. KEY WORDS	LINK A		LINK B		LINK C	
	ROLE	WT	ROLE	WT	ROLE	WT
Reactor vessels						
Structural integrity						
Radiation damage surveillance						
Reactor structural steels						
Transition temperature						
Mechanical constraint						
Fracture safe design						
Fracture transition elastic						
Fracture toughness						
Gradient effects						
Impact shelf toughness						
Ratio analysis diagram						
Graphical fracture mechanics						

DD FORM 1 NOV 65 1473 (BACK)
(PAGE 2)

Security Classification

CONTENTS

ABSTRACT

The structural implications of radiation effects to nuclear reactor pressure vessels are assessed primarily through surveillance programs in which the properties of the vessel are projected from an evaluation of small specimens of the vessel steel. In the USA, the current fracture-safe criterion requires that the vessel operating temperature, at certain stress levels, be at the FTE (Fracture Transition Elastic) temperature, defined as NDT+60°F(33°C), derived from surveillance measurements. Review of available data from five reactor surveillance programs indicates that this criterion is adequate for the vessels concerned. Complete assurance of fracture-safe operating conditions can be attained through a limit-analysis procedure that considers and integrates the effects of five factors: (a) the radiation-induced shift in transition temperature, (b) the initial shelf energy, (c) the radiation-reduced ductile shelf energy, (d) the effects of the fluence (and toughness) gradient through a thick vessel wall, and (e) the effects of thickness-induced mechanical constraint.

PROBLEM STATUS

This is a final report on one phase of the problem; work on other phases is continuing.

AUTHORIZATION

NRL Problem M01-14
Projects RR022-11-41-5409 and AEC-AT(49-5)-2110

Manuscript submitted October 6, 1971.

PROCEDURES FOR INTERPRETING THE STRUCTURAL IMPLICATIONS OF RADIATION-DAMAGE SURVEILLANCE RESULTS ON NUCLEAR PRESSURE VESSELS

INTRODUCTION

In water-cooled commercial nuclear power plants, the neutron radiation-induced embrittlement of the primary containment vessel is assessed through a surveillance program. This program consists of a long-range plan for periodically removing and evaluating the properties of metallurgical specimens representative of the vessel, as fabricated. Interpretations of these data form the basis for assuring a safe continuing reactor operation. Generally, only limited surveillance data are available because of the complications introduced by irradiating specimens in an operating reactor plant. Consequently, every effort must be made to examine these data in terms of the latest developments in fracture-safe design analysis. This report examines fracture analysis procedures in terms of their relationship to the growing volume of data from surveillance programs.

It should be noted that there are two possible bases for the fracture-safe design of any structure. These are:

a. *Nonprovisional*—based on assurance that unstable fracture cannot evolve. In simple terms, this approach guarantees the prevention of fracture extension at elastic stress levels irrespective of flaw size.

b. *Provisional*—based on calculation of critical flaw size-stress combinations for the initiation of fracture at elastic stress levels. That is, the safety of the structure is provisional to (dependent on) not exceeding such critical combinations of flaw size and stress.

The provisional method requires exact definition of the degree of embrittlement of all component parts of the reactor pressure vessel with high statistical confidence. Further, it requires exact definition of flaw sizes and stress levels. These multiple requirements are complicated by statistical variations of all relevant factors. Thus, engineering prudence requires establishing criteria based on the "worst" possible combinations of metal sensitivity to radiation, flaw size, and stress levels. The net result is that "lower bound" values of fracture resistance are established which are essentially equal to the lower limit set by the nonprovisional criteria. The procedures by which the "lower bound" values are established are complex and difficult to document by means of limited surveillance specimens of small size.

The nonprovisional approach also sets a "lower bound" value for fracture resistance based on ensuring that the metal is not frangible, i.e., fracture can occur only by exceeding yield stresses. The procedures for establishing and documenting this desirable condition are relatively simple because the complexities of statistical flaw size-stress relationships are eliminated from consideration.

Most importantly, the provisional approach suffers from the tacit assumption that the pressure vessel is operated within the frangible (brittle) range. If this is not the case, then there is no basis for using the provisional method. In the event of a misjudgment of flaw size and stress levels, leading to a brittle fracture, the fact that the pressure vessel was knowingly operated in the brittle range would be of catastrophic consequences as to legal, professional, and public confidence factors.

For these various reasons the authors firmly abide by the guiding principle that reactor operations should be based on credible evidence that the reactor metal is operating in an environment above the brittle-to-ductile transition region, thereby eliminating the need for the provisional approach. All discussions of this report center on this principle and on means for guaranteeing that this desirable state of the metal is retained both as to the transition temperature and shelf energy factors.

CURRENT FRACTURE ANALYSIS CRITERION

The currently accepted criterion for assuring high fracture toughness depends upon the fact that the steels currently used for light-water reactor vessels exhibit a marked transition from brittle-to-ductile fracture behavior over a relatively narrow temperature range. In service, neutron irradiation elevates this brittle-ductile transition, thus lowering the toughness and increasing the strength (1). The condition of the irradiated vessel is assessed in service by exposing a series of notched impact bars (usually Charpy V-notch) (2) and tension specimens of the vessel steel (plate, or forging, and the related weld and weld heat-affected zone (HAZ) materials) at the vessel wall for periodic removal and evaluation. The data from the surveillance program are interpreted by means of the Fracture Analysis Diagram (FAD) shown in Fig. 1 (3). The primary reference point of the FAD is the Drop Weight-Nil-Ductility Transition (DW-NDT) temperature (4) which denotes the point at which the brittle-ductile transition begins. At this temperature, very small flaws can become unstable at yield stress levels, while larger flaws require proportionately lower levels of elastic stress for propagation, as seen in the figure. In the course of the transition, a temperature is reached at which even very large flaws require stresses in excess of yield to propagate. This temperature is called the Fracture Transition Elastic (FTE) temperature and is defined as being equal to NDT+60°F(33°C). The current fracture safety criterion, endorsed by the U. S. Atomic Energy Commission (USAEC), restricts vessel operating temperatures to those above the FTE temperature if stresses exceed some established limit. This criterion provides assurance of fracture safety with respect to a shift in transition temperature. In fact, the criterion may be conservative since it is applied only to the highly irradiated inside surface of the vessel, while the remainder of the vessel wall in the direction away from the core is progressively more ductile.

It is the purpose of the surveillance program to assess the irradiation-induced elevation of the FTE temperature throughout the lifetime of the vessel. The Charpy-V specimen is used to establish the NDT temperature and, therefore, the FTE temperature by means of an "energy-fix" correlation. Also, the Charpy-V specimen yields additional information as to shelf level energy at the ductile end of the transition, as discussed later. Drop-weight specimens are not routinely used to establish the irradiation-condition NDT temperature because they are too large for ready inclusion in surveillance programs.

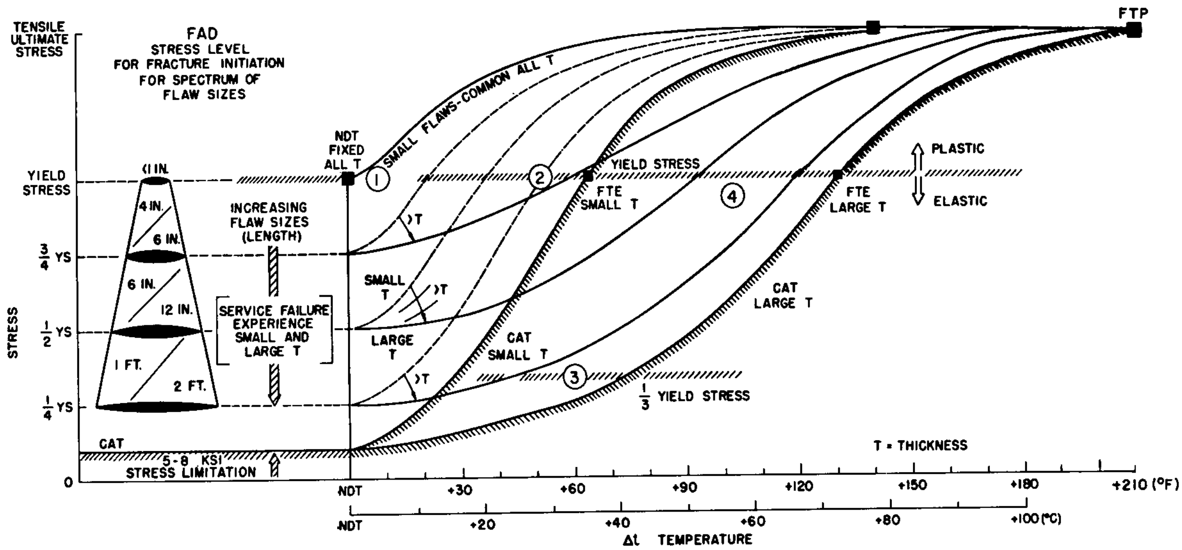

Fig. 1—Generalized Fracture Analysis Diagram (FAD) as referenced by the NDT temperature (see Ref. 3).

In summary, to adequately cover the ΔFTE aspects of the irradiated vessel, surveillance analysis depends upon:

a. a preirradiation correlation of a Charpy-V energy value with the NDT for the materials (plate or forging, weld, and weld HAZ) actually used in the vessel beltline or highest fluence region;

b. the determination of the radiation-induced transition shift (based upon Charpy-V tests) for the highest fluence and the most radiation-sensitive surveillance steel; and

c. the addition of a temperature factor which assures operation at FTE or higher while certain stress limits are applied.

This approach is sufficient for most projectable reactor vessel conditions.

DEVELOPMENTS WHICH IMPROVE FRACTURE ANALYSIS AND SURVEILLANCE DATA INTERPRETATIONS

Introduction

Recent developments in fracture analysis have provided additional facts which should be considered in evaluating the structural significance of the typically limited surveillance data. These developments consider the fluence and toughness gradient in thick-walled vessels, the effects of mechanical constraint associated with thick sections, and the effect of radiation-reduced ductile shelf level.

Gradient Effects

The fracture safety criterion described above is applied as though the whole vessel were uniformly affected by a radiation exposure equal to that on the inside vessel wall. Figure 2 describes the true pattern of neutron fluence attenuation in an 8-in. vessel wall (5). This curve was derived from spectrum calculations of a number of real vessels, as well as from through-thickness measurements made in vessel-wall-simulation experiments (5,6). Note that in the first 2 in. (50 mm) the fluence drops more than 50 percent. The advantage of this reduction in terms of embrittlement (or radiation damage) will depend on the various factors which influence the degree of steel embrittlement, especially the steel composition, the service temperature, and the fluence range involved (surface to 2 in. deep). Nevertheless, the advantage is clear. At the lower fluences ($\leqslant$ about 1.5×10^{19} n/cm^2 for neutron energies >1 MeV) where embrittlement is approximately linear with incremental fluence additions, embrittlement 2 in. beneath the inner vessel wall will be only about one-half that at the inner vessel wall. Similar advantages accrue on a progressive basis through a thick-walled vessel. Higher steel ductility at deeper locations in the vessel wall means both a progressively smaller radiation-induced ΔFTE and a progressively smaller reduction in the ductile shelf energy. Hence, application of the criterion which requires maintenance of the stressed vessel at temperatures above FTE=NDT+60°F for the inner wall is conservative for most cases. A means for relaxing this criterion after several years of irradiation service seems desirable. A suggestion in this direction proposed by the authors is to use the irradiated condition existing at the one-quarter thickness ($T/4$) location for applying a thickness-adjusted temperature limit criterion. However, it is beyond the scope of this report to develop this concept further.

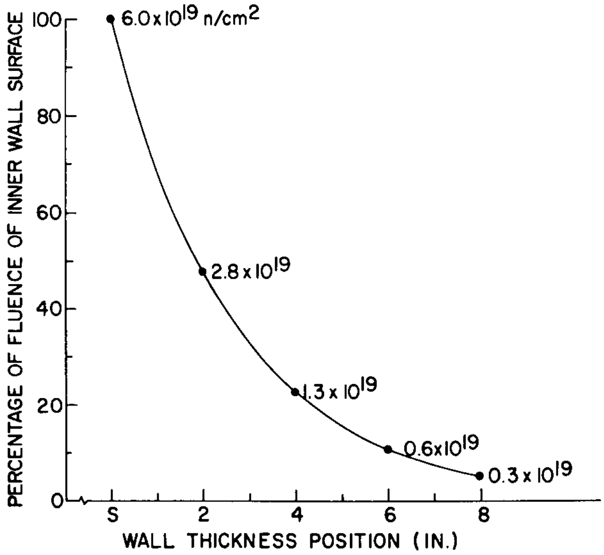

Fig. 2—Fluence attenuation through an 8-in. pressure vessel wall expressed as a percentage of the fluence received by the inner wall surface nearest the fuel core. Fluence values are listed along the curve.

Thickness Effects

A consideration which tends to offset the conservatism mentioned above is the effect of thickness-induced mechanical constraint. Recent investigations have defined this effect for A533-B steel. These investigations centered on linear elastic fracture mechanics (LEFM) methods at the Westinghouse Corp. (7) and dynamic tear (DT) tests at NRL (8). The results showed that the effect of thickness is to expand the temperature range in which the brittle-ductile transition occurs. Nevertheless, the temperature-induced micrometallurgical processes eventually overcome the effects of mechanical constraint, and a sharp transition with a ductile "shelf" condition is attained, as is the case with thin sections of the same steel.

The effects of constraint are incorporated in a revised Fracture Analysis Diagram (FAD) (9) in which the FTE for section thicknesses of 6 to 12 in. (150 to 300 mm) is elevated by about 70°F (39°C) on the basis of the NRL large-specimen DT investigation. Thus, a criterion of NDT+130°F(72°C) may be appropriate for the unirradiated state, but when coupled with radiation-induced shifts in transition temperature (ΔT), it is overly conservative for a vessel which necessarily exhibits the toughness gradient described above. Therefore, it is suggested that NDT+60°F(33°C) is sufficient for an irradiated vessel because of the advantages of a "composite" vessel in which the tough outer portions will control full-section behavior. (See Ref. 5 for a detailed discussion of this factor.)

One approach which recognizes the toughness gradient state would be to apply the suggested thickness-adjusted FTE for the $T/4$ irradiated-vessel condition. However, the added factor—radiation reduction of fracture resistance (shelf drop)—also must be considered.

Shelf Reduction

The philosophy behind the FAD (or FTE) approach hinges on the ability of the steel to develop a relatively high shelf level toughness. This means a value higher than 50 ft-lb in an unirradiated state. If a vessel steel is not capable of developing needed shelf toughness because of initial properties, directionality, cleanliness, radiation exposure, or a combination of these, then shelf toughness may become the limiting factor rather than the FTE. Under these circumstances, raising the temperature has no effect. That is because the steel's toughness will not be enhanced by higher temperatures, and less-than-yield-point loading may cause fracture in the presence of a sufficiently large flaw regardless of temperature. The range of shelf-reduction values is illustrated in Fig. 3 which shows the Charpy-V shelf values versus radiation-induced transition temperature increases ΔT for a number of steels irradiated at about 550°F (288°C) to various fluence levels, both in reactor surveillance programs and in test reactors. The shelf level values obtained to date are all relatively high even after rather severe irradiation. To this must be added the gradient advantage which is quite significant in the irradiated case.

The breadth of the data band (Fig. 3) is indicative of the variable shelf response of reactor vessel steels regardless of copper content (and even phosphorus and sulfur content). This point is especially important in view of the fact that published plots (10,11) of percent shelf drop ($\Delta E/E$) versus ΔT have suggested an exceptionally consistent and precise relationship. Such a precise trend band would be of high value for establishing a boundary (worst case) for reactor operating limits. However, the variability shown in Fig. 3 for several typical

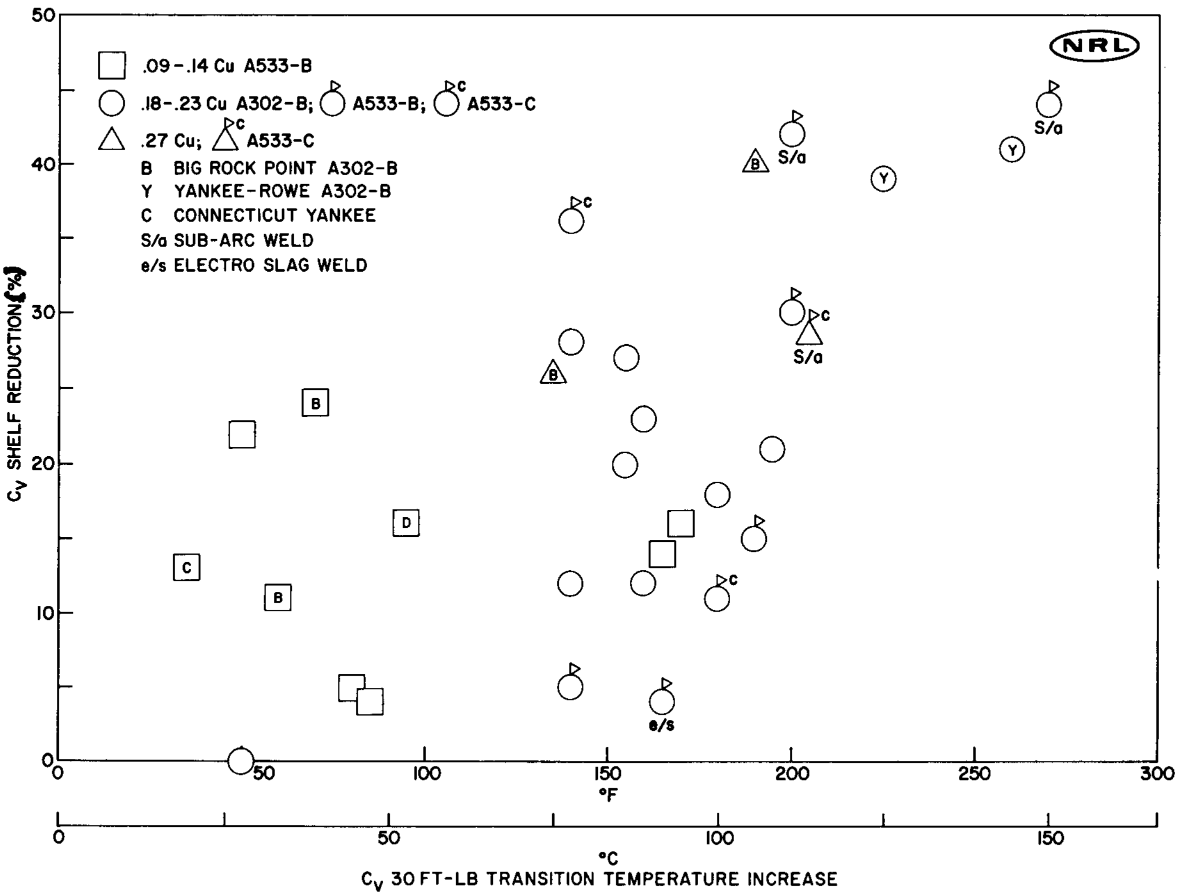

Fig. 3—Percent reduction in Charpy-V shelf energy vs transition temperature increase for pressure vessel steels irradiated at ~550°F (288°C). No consistent trend is evident from steel type, strength level, or chemical composition.

A302-B and A533-B vessel steels effectively demolishes this premise. Thus, the only way to know this limiting (weak link) value is to measure it.

Having the limiting shelf level values for the irradiated vessel, the question becomes one of how to extend these data to analyze the potential for structural integrity of the structure.

The meaning of low shelf energy as determined by a Charpy-V test is difficult to define quantitatively in terms of structural performance. The LEFM techniques can be applied, for example, but only if the shelf is reduced to the point of frangibility. Validation using irradiated LEFM specimens is quite unlikely, however, because of the large specimen size that would be required. The only approach which currently holds an answer to this requirement is the Ratio Analysis Diagram (RAD) (12) which provides a graphical means of projecting fracture toughness at shelf level temperatures. This technique also permits the integration of strength transition (the factor of generally reduced toughness with increasing strength), which is important in analyzing the structural potential of radiation-embrittled steels.

The RAD, as shown in Fig. 4, is a simple graphical presentation showing the interrelations between shelf level DT energy, shelf level K_{Ic} data, and yield strength. Entry into the

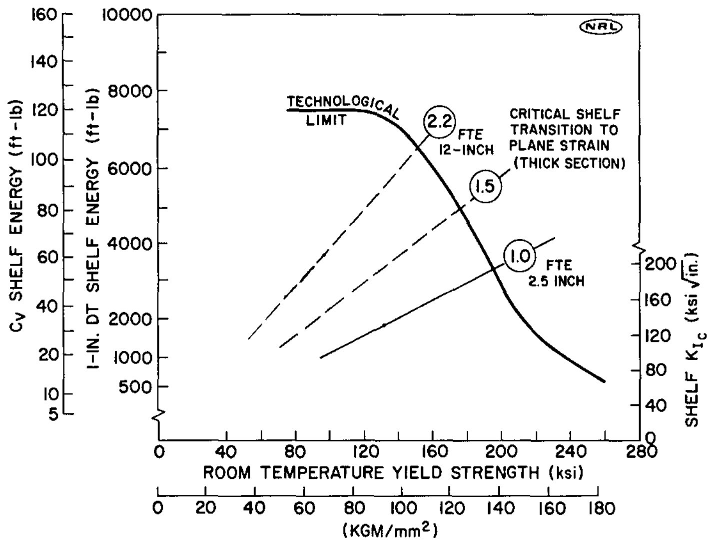

Fig. 4—Example of a Ratio Analysis Diagram (RAD). The grid of K_{Ic}/σ_{ys} ratio lines (circled in the diagram) indexes the region of the diagram which relates to plane strain fracture for steels of a given thickness.

diagram is accomplished with yield strength data coupled with either DT or K_{Ic} values. The location of the point for a steel provides a graphical basis for assessing the relative toughness quality and, hence, the service performance to be expected. A detailed description is presented by Pellini (13).

The RAD was developed from correlations of DT and K_{Ic} shelf values for high-strength steels. A correlation (14) of the Charpy-V to DT (before and after irradiation) provided the entry scale for Charpy-V shelf level data. The yield strength scale provides a means for placement of the K_{Ic}/σ_{ys} constant ratio lines. These ratio lines provide a key index for measuring the fracture resistance of a metal. Each ratio line corresponds to a specific section thickness; the ratio 1.0 corresponds to 2.5 in., the ratio 2.2 corresponds to 12 in., etc., as derived from

$$B \geqslant 2.5(K_{Ic}/\sigma_{ys})^2 \tag{1}$$

where B is the section thickness (15). If a point entered on the RAD falls above the ratio for the section thickness of interest, behavior above FTE (tough) is projected. If the data point falls below the ratio line, frangible behavior is projected. Movement upward along one of the ratio lines means that for increasing yield strength, the DT or K_{Ic} shelf value must also increase if equivalent fracture toughness is to be maintained at the higher strength. If shelf level values remain constant or decrease with increasing yield strength, a point representing the material in question may move, for example, from a position of relatively high toughness (above the ratio line for the thickness) to much lower toughness, possibly even frangibility, if the point falls below the line.

A complementary analysis diagram to the RAD permits determination of the critical flaw size for a given section thickness as a function of the imposed stress level and of the K_{Ic}/σ_{ys} ratio. This is called (9) the Graphical Fracture Mechanics (GFM) plot, as shown in Fig. 5. Entered is the condition for a 10-in.-thick section having a ratio of 2.0; at (1/2) σ_{ys}

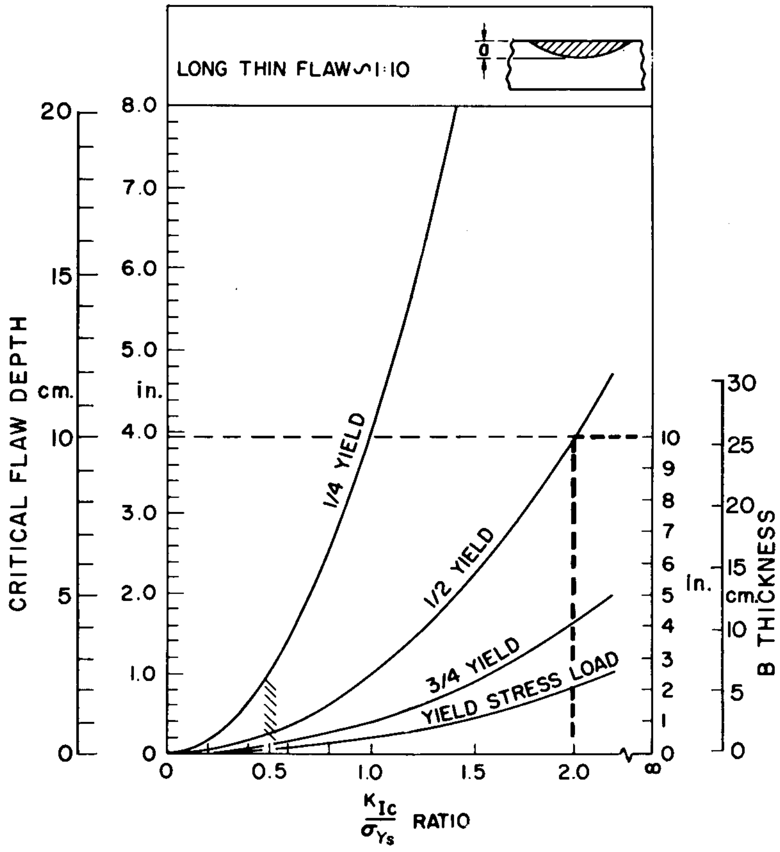

Fig. 5—Graphical relationship of flaw size and stress requirements for the initiation of plane strain fracture as a function of increasing K_{Ic}/σ_{ys} ratios for a long thin crack. The section size requirements are indicated by the B scale (9).

the critical flaw size for a 10:1 long thin flaw is almost 4 in. The plot thus provides a graphical solution to the equation

$$K_{Ic} = \frac{1.1}{\sqrt{Q}}\sqrt{\pi a} \tag{2}$$

for plane strain initiation from a semielliptic surface flaw for a plate in tension. Here, a is the crack depth, the shape factor Q is assumed, and the highest ratio of K_{Ic}/σ_{ys} is known for the steel thickness of interest. A hypothetical example of a surveillance situation is provided later in this report to show how these diagrams may be used for the analysis of irradiated steel. Before that, however, an example of current surveillance practice and its use in operating a reactor is provided in the next section for reference.

CURRENT SURVEILLANCE ANALYSIS PRACTICE

Pressure vessel surveillance programs currently being implemented are in general conformance with the ASTM E-185 standard, Recommended Practice for Reactor Vessel Surveillance (16). This calls for surveillance capsules containing Charpy-V and tension test specimens machined from base metal, weld metal, and the weld HAZ; inclusion of additional specimens, such as the fracture mechanics wedge opening loading (WOL) or compact tension (CTS) is also permitted. All pressure vessel surveillance programs in the USA have included Charpy-V and tensile specimens and, while base metal was always included, weld metal and weld HAZ specimens were not in every case.

A typical comprehensive program of the early reactors is that of the Big Rock Point Boiling Water Reactor. The surveillance program (17) provided for irradiation in accelerated (shield) and vessel-wall locations, as well as for thermal exposure in out-of-flux thermal-control locations, of base metal, weld metal, weld HAZ, and "standard" reference metal Charpy-V and tension specimens.

The testing and evaluation of a series of capsules withdrawn from all three types of exposure locations have been described (18). The Charpy V-notch ductility results from the weld metal specimens, for example, are shown in Fig. 6 (18); here the primary effort was the accurate definition of the increase in 30 ft-lb transition temperature. The surveillance program also provided shelf energy absorption levels and tensile results; however, no use has been made of these data other than to note the general trends with irradiation.

The typical end result of surveillance analysis is formulation of a trend for increasing transition temperatures versus neutron fluence, such as shown in Fig. 7 (18). The weld metal can be seen as the obvious limiting material for this reactor vessel. Using the neutron fluences calculated for the service lifetime of the plant, the maximum transition temperature increase, and also the actual as-irradiated weld metal, the transition temperature ΔT can be projected. For the Big Rock Point weld metal, the ΔT was 250°F (139°C) so that, allowing for the initial −70°F (−57°C) transition temperature, the final weld metal transition temperature was projected to be 180°F (82°C). Based on this type of analysis, the Big Rock Point pressure vessel would be considered fracture safe for operation over its intended 40-yr lifetime (18).

STRUCTURAL SIGNIFICANCE OF SURVEILLANCE DATA

Consideration of the effects of radiation (supplied by the surveillance program), coupled with the gradient, mechanical constraint, and shelf reduction factors, forms the basis for projecting the structural significance of surveillance data. It is clear that the established criterion which requires a vessel temperature equal to NDT+60°F (33°C) is an adequate technique since positive gradient effects can be expected to more than compensate for the constraint effects. The question of ductile shelf reduction, however, requires analysis for the individual vessel "weak link" material, plus analysis by a technique such as the RAD, coupled with a graphical or computed fracture mechanics integration of stress and flaw size factors for the thickness of the vessel under consideration.

Fortunately, no specific example of frangibility due to radiation-induced reduction in shelf energy is presently known, but an examination of selected data (such as that shown in Fig. 8 and Table 1) makes it possible to postulate such a condition. The necessary conditions for frangibility are that the decrease in shelf energy and the corresponding increase in yield strength combine to index a point below the ratio of K_{Ic}/σ_{ys} on the RAD for the thickness of the vessel steel. Consider the measured surveillance data points plotted on the RAD in

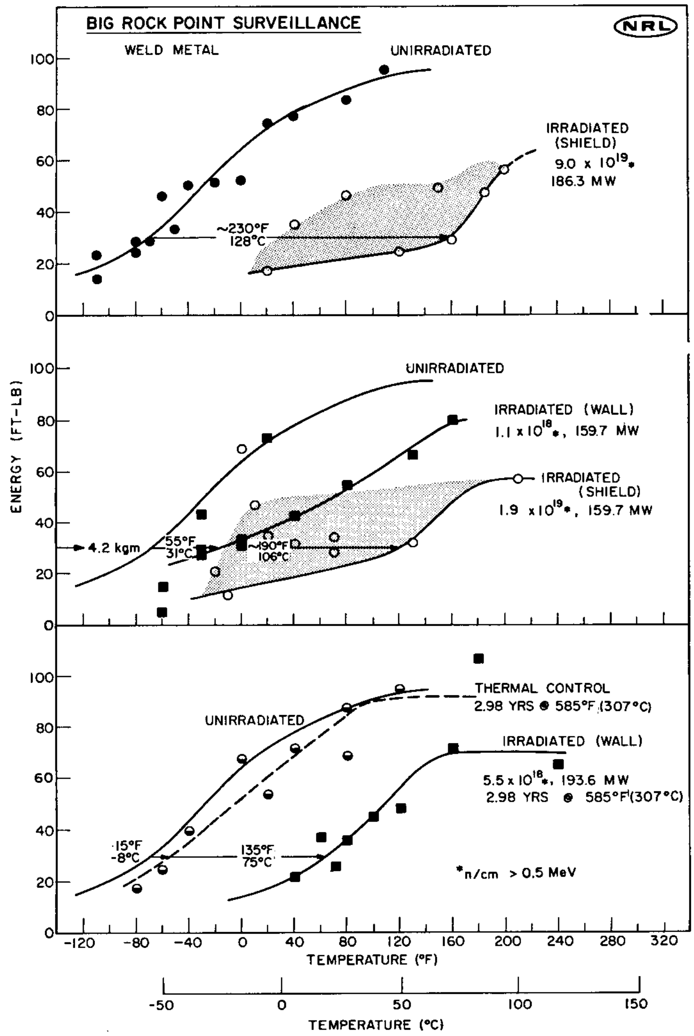

Fig. 6—Notch ductility characteristics of weld metal Charpy-V specimens irradiated in the Big Rock Point pressure vessel surveillance program. For reference comparisons, fluences > 0.5 MeV are converted to n/cm^2 > 1 MeV (fission spectrum) by cross-section ratios of 118/98 for accelerated (shield) locations and 126/98 for vessel-wall locations.

Fig. 8. (The details for the surveillance data plotted are given in Table 1.) The two points for the Yankee-Rowe reactor are derived from accelerated irradiation damage surveillance positions which produced fluences of 5×10^{19} and 7×10^{19} n/cm^2 >1 MeV. These fluences are well in excess of the projected 2.5×10^{19} n/cm^2 >1 MeV Yankee pressure vessel lifetime (19,20). The single point (first capsule result) for the Connecticut-Yankee plant represents a fluence of only 2.1×10^{18} n/cm^2 >1 MeV (21).

Since both the Yankee-Rowe and the Connecticut-Yankee vessels are fabricated from heavy-section A302-B steel, it is fair to suggest that they could represent A302-B steel behavior in power reactor vessels; in fact, note that the data (Fig. 8) do follow the trend band established for the 6-in. A302-B steel reference plate (RW to WR) (22). Thus, if some hypothetical heavy-section pressure vessel fabricated from A302-B steel featured a relatively low

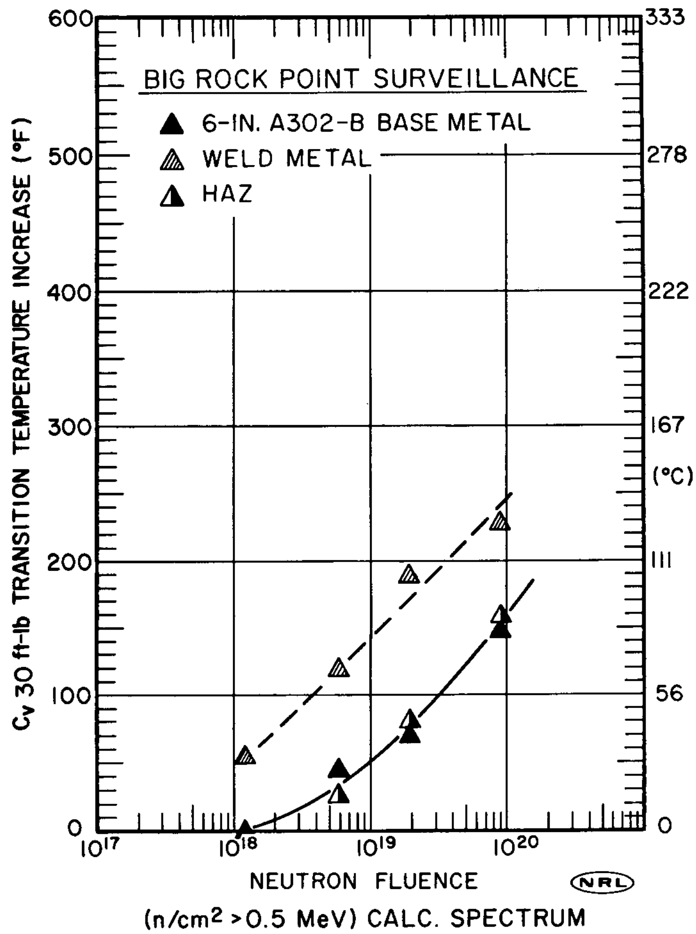

Fig. 7—Increase in Charpy-V 30-ft-lb transition temperature neutron fluence for Big Rock Point surveillance materials. The weld metal shows an increase of 250°F (139°C) for the 40-yr fluence of 8.1X10^{19} n/cm^2 (>0.5 MeV) or (1.0X10^{20} n/cm^2 (>1 MeV) fission spectrum) (18).

initial shelf level and reached an ultimate service limit fluence in the range of 5-7X10^{19} n/cm^2 >1 MeV, it is quite reasonable to project a decrease in shelf similar to that depicted by the trend band of the Yankee-Rowe points in Fig. 8. Note that these points fall in the range of the line for a ratio of 1.5, corresponding to a thickness B of 5.6 in. This means that such a vessel, about 6 in. (150 mm) in thickness, could exhibit frangible behavior. This corresponds with approximately 50-ft-lb (6.9-kgm) Charpy-V shelf levels. Note further that if the hypothetical vessel thickness were 10 in. (or 12 in.), the K_{Ic}/σ_{ys} ratio would have to exceed 2.0 (or 2.2), and the related Charpy-V shelf levels would have to be proportionately higher to preclude fracture at stresses lower than yield. The shelf level yield strength data point for the hypothetical vessel would fall directly into this range, and fracture initiation might lead to unstable fracture if the vessel were uniform in toughness. This, fortunately, is not the case to be expected for a vessel of high initial shelf value since the steep gradient in fluence described in Fig. 2 would lead to progressively higher shelf level toughness at greater depths from the inside vessel wall. For example, it is reasonable to project that a radiation-induced shelf drop of 20 percent at the inner wall might be only 10 percent at the $T/4$ location.

Table 1
Mechanical Property Change Data from Selected Pressure Vessel Surveillance Programs in the USA

Reactor	Type of Steel	Steel Form	Neutron Fluence (10^{19} n/cm^2 >1 MeV)	Charpy-V Shelf Energy (ft-lb)		Room Temperature Yield Strength (ksi)	Charpy-V 30 ft-lb Transition Temp. ΔT		Surveillance Location
				Pre	Post		°F	°C	
Big Rock Point	A302-B	Plate* ○	0.71	82	73	84	60	33	Vessel Wall
		Plate ○	10.7	82	70	89	230	128	Accelerated
Dresden I	A302-B	Plate □	0.98	73	58	73.5	95	53	Inside Wall
		Plate □	5.5	73	50	83	150	83	Near Core
		HAZ ◨	13.0	70	52	85	145	81	Near Core
Humboldt Bay	A302-B	Weld ⧨	2.5	88	68	80.5	105	58	Accelerated
		HAZ ⧩	2.5	60	45	79	75	42	Accelerated
Connecticut-Yankee	A302-B	HAZ ⬡	0.21	88	63	70 to 80	70	39	Accelerated
Yankee-Rowe	A302-B	Plate △	5.0	76	46	103.2	225	125	Accelerated
		Plate △	7.0	76	45	120	260	144	Accelerated
PM-2A	A350-LF3	Plate ◇	0.73	95	66	92.2	195	108	$T/4$, Vessel

*Symbols correspond to symbols on Fig. 8.

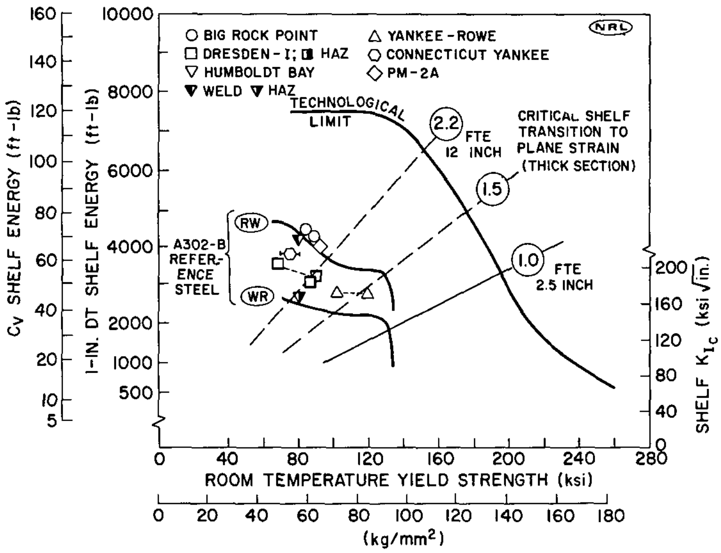

Fig. 8—Ratio Analysis Diagram for steels showing trend lines for A302-B steel and measured Charpy-V shelf yield strength values from surveillance programs. All open symbols represent base plate. (See Table 1 for specific data.)

The critical flaw size (for long thin flaws) for unstable fracture under these hypothetical conditions, at one-half yield stress, are in the range from 3 in. deep by 30 in. long (7.6:76.0 cm) for the 8-in. vessel, to 4 in. deep by 40 in. long (~10:100 cm) for the 10-in. thick vessel (Fig. 5). Higher local stresses would require smaller flaws, but in the most highly irradiated portions of the vessel, higher stresses are unlikely.

SUMMARY AND CONCLUSIONS

Reactor vessel radiation damage surveillance data are always limited. Under these circumstances optimum use must be made of available data in order to project the structural implications of radiation damage to the steel vessel. Furthermore, since all the data are based on relatively small laboratory test specimens, the procedures for interpreting the surveillance data in terms of the reactor vessel structure are most important. For these reasons, recent advances in fracture analysis have been reviewed within the context of available data from reactor surveillance programs.

Four factors must be considered: (a) the radiation-induced increase in transition temperature, or, more accurately, the increase in the Fracture Transition Elastic (FTE) temperature, (b) the toughness gradient produced by neutron attenuation in the thick vessel, (c) the effect of vessel thickness in evaluating the FTE, and (d) the reduction in fracture energy and increase in yield strength with neutron exposure. As currently applied in the USA, the criterion for assuring vessel integrity depends upon item (a) alone, that is, maintaining the vessel temperature above the irradiated FTE temperature. This is accomplished by measuring the NDT, finding a correlation point on the Charpy-V curve before irradiation, translating the NDT by a temperature equal to the shift observed by tests of the surveillance specimens, and adding 60°F (33°C) to assure an FTE during the pressurization or operating cycle of the reactor. This procedure is applied to conditions at the inner vessel wall so that no advantage is taken of the progressively tougher steel at points through the vessel. On the other hand, no penalty is taken for thickness-induced mechanical constraint elevation of FTE. These two factors may be considered as compensating so that the long-standing criterion of NDT+60°F still represents a conservative approach.

Analysis of the implication of shelf energy and its reduction suggests that initial shelf values should exceed 50 ft-lb if the radiation levels are such as to significantly reduce shelf energy. If this initial condition is met, advantage may be taken of the toughness gradient because the "composite" irradiated vessel wall will be far superior to the inner surface shelf situation as defined by the surveillance program. The only available analysis technique for determining the minimum Charpy-V energy level for assured vessel integrity is the Ratio Analysis Diagram (RAD). This diagram permits the evaluation of irradiated fracture toughness in terms of shelf level Charpy-V and yield strength, or shelf level K_{Ic} and yield strength. This two-way entry, plus the knowledge that with an increase in the yield strength a higher toughness is required for a given thickness, permits a realistic analysis for a given reactor vessel situation. Considering the toughness gradient, shelf drops to values of 50 ft-lb are acceptable for the inner vessel wall. Reductions to below the value of 50 ft-lb for the inner wall would signal a situation where the toughness gradient might not be sufficient to assure against unstable fracture.

The data from several reactor surveillance programs have been analyzed within the framework of these new developments in fracture analysis, and the following conclusions may be stated:

(a) Only in the case of a highly radiation-sensitive vessel steel and a relatively high neutron flux at the reactor vessel wall is the current criterion (operation above FTE) likely to require additional consideration in operational procedures.

(b) The application of this criterion to the inner vessel wall condition is conservative because of steep neutron fluence gradients which produce important toughness gradients that suggest much greater "composite" vessel toughness than is implied by surveillance data.

(c) Temperature adjustments for thickness constraint effects in the context of the ΔT (or ΔFTE) do not usually change the conclusion of item (a) above. In fact, the gradient described in item (b) more than compensates for this factor.

(d) Vessels in which the base metal or weld constituents have low initial Charpy-V shelf values and to which relatively high neutron irradiation exposures accrue may produce a situation requiring either shelf level analysis for critical flaw sizes for given stress levels, or provision for correcting the radiation damage, or removal from service. Hypothetical examples of shelf reduction situations have been cited, though no real situation where this factor is critical can be cited.

(e) Thickness has implications not only in the ΔFTE context but also in response to shelf reduction since, with increasing thickness, the K_{Ic}/σ_{ys} ratio line lies progressively higher on the Ratio Analysis Diagram. Thus, for greater thickness, higher minimum shelf values are required to assure FTE behavior.

(f) Further study is required in order to develop quantitative relationships between radiation-induced shelf drop, yield strength increase, and toughness factors. In the meantime the accepted concept of NDT+60°F (33°C), coupled with a weak-orientation initial shelf limit of 50 ft-lb and a shelf drop which does not fall below this value for the inner vessel wall, should be considered conservative practice.

By integrating the best knowledge of fracture-safe analysis procedures with the best possible analysis of the always-limited reactor vessel surveillance data, it is possible to accurately project the long-range condition of comercial water-reactor pressure vessels and thereby make the decisions necessary for assuring fracture safety. Several additional factors also support the probabilities for safe reactor containment. These include the improved steels now being produced (the Type A533 steels usually exhibit quite high initial shelf values), and the reduction of radiation sensitivity by control of residual elements in steels now being produced. (Supplemental specifications requiring control of residual elements at certain maximum levels are now being considered by the American Society for Testing and Materials and by the American Welding Society.)

ACKNOWLEDGMENTS

The research described was sponsored by the U.S. Atomic Energy Commission, Division of Reactor Development and Technology, Fuels and Materials Branch, and by the Office of Naval Research. The support of these organizations is acknowledged with gratitude. Discussions with Messrs. W.S. Pellini and F.J. Loss relative to fracture analysis procedures and their interpretation were invaluable and are gratefully acknowledged. Thanks are due to Mr. T.D. Keenan, Yankee Atomic Electric Company, for the initial surveillance report and for discussions regarding the Connecticut-Yankee reactor vessel. The devoted efforts of the members of the Reactor Materials Branch in evaluating many of the irradiated specimens are also acknowledged with sincerest appreciation.

REFERENCES

1. Steele, L.E., "Neutron Irradiation Embrittlement of Reactor Pressure-Vessel Steels," *Atomic Energy Rev.*, VII(No. 2):122-132, IAEA, Vienna 1969.
2. "Standard Methods for Notched Bar Impact Testing of Metallic Materials, ASTM Designation E23-66," *ASTM Standards*, Part 31, pp. 271-285, July 1970.
3. Pellini, W.S., Steele, L.E., and Hawthorne, J.R., "Analysis of Engineering and Basic Research Aspects of Neutron Embrittlement of Steels," NRL Report 5780, Apr. 1962; also, *Welding J.*, 41(No. 10):455-s-469-s, 1962.
4. "Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels, ASTM Designation E208-69," *ASTM Standards*, Part 31, pp. 582-601, July 1970.
5. Loss, F.J., Hawthorne, J.R., Serpan, C.Z., Jr. and Puzak, P.P., "Analysis of Radiation-Induced Embrittlement Gradients on Fracture Characteristics of Thick-Walled Pressure Vessel Steels," NRL Report 7209, Mar. 1971; also, *Trans. ASME J. Eng. for Ind.*, Series B, 1971.

6. Serpan, C.Z., Jr., and Watson, H.E., "Through-Thickness Neutron Fluence and Embrittlement Gradients in Reactor Pressure Vessels," Nuclear Technol., 11(No. 4): 592-601, Aug. 1971.

7. Wessel, E.T., "Linear Elastic Fracture Mechanics for Thick-Walled, Welded Steel Pressure Vessels: Material Property Considerations," in *Practical Fracture Mechanics for Structural Steel*, pp. H1-H44, London:Chapman and Hall, 1969.

8. Loss, F.J., "Dynamic Fracture Characteristics of Thick Section Steels," ASME-AIME Symposium on The Technology of Pressure Retaining Steel Components, Vail, Colorado, Sept. 21-23, 1970; also, *Nuclear Met.*, 16:98-118.

9. Pellini, W.S., "Evolution of Engineering Principles for Fracture-Safe Design of Steel Structures," NRL Report 6957, Sept. 1969.

10. Burghard, H.D., and Norris, E.B., "The Effect of Neutron Irradiation on the Fracture Characteristics of Pressure Vessel Steels," ASME Paper 67-Met-1 presented at ASME Metals Engineering Conference, Houston, Texas, April 3-5, 1967.

11. Kawasaki, M., et al., "Evaluation of the Embrittlement of Pressure Vessel Steels Irradiated in JPDR" (Japan Power Demonstration Reactor); also, "Irradiation Effects on Structural Alloys for Nuclear Reactor Applications," ASTM Special Tech. Publ. 484, pp. 74-94, Disc., p. 95, 1971.

12. Pellini, W.S., "Advances in Fracture Toughness Characterization Procedures and in Quantitative Interpretations to Fracture-Safe Design for Structural Steels," NRL Report 6713, Apr. 1968; also, Welding Res. Council Bull. No. 130, May 1968.

13. Pellini, W.S., "Integration of Analytical Procedures for Fracture-Safe Design of Metal Structures," NRL Report 7251, Mar. 1971.

14. Loss, F.J., Hawthorne, J.R., and Serpan, C.Z., Jr., "A Reassessment of Fracture-Safe Operating Criteria for Reactor Vessel Steels Based on Charpy-V Performance," NRL Report 7152, Sept. 1970; also, Trans. ASME J. Basic Eng., Series D, 1971.

15. "Tentative Method of Test for Plane-Strain Fracture Toughness of Metallic Materials, ASTM Designation E399-70T," *ASTM Standards*, Part 31, pp. 911-927, July 1970.

16. "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels, ASTM E185-70," *ASTM Standards*, Part 30, pp. 442-446, July 1970.

17. Brandt, F.A., "Reactor Pressure Vessel Material Surveillance Program at the Consumers Power Company, Big Rock Point Nuclear Plant," Report GECR-4442, General Electric Company, Dec. 1963.

18. Serpan, C.Z., Jr., and Watson, H.E., "Mechanical Property and Neutron Spectral Analyses of the Big Rock Point Reactor Pressure Vessel," Nuclear Eng. and Design, 11(No. 3):393-415 (1969; published Apr. 1970).

19. Serpan, C.Z., Jr., and Hawthorne, J.R., "Yankee Reactor Pressure Vessel Surveillance: Notch Ductility Performance of Vessel Steel and Maximum Service Fluence Determined from Exposure During Cores II, III, and IV," NRL Report 6616, Sept. 1967.

20. "Yankee Plant—Fast Neutron Exposure, Transition Temperature Increase and Operating Limitations for the Reactor Vessel," Report RFS-M-5, Yankee Atomic Electric Company, June 1966.

21. Ireland, D.R., and Scotti, V.G., "Examination and Evaluation of Capsule A for the Connecticut-Yankee Reactor Pressure-Vessel Surveillance Program," Battelle Memorial Institute, Columbus, Ohio, Oct. 30, 1970.

22. Hawthorne, J.R., "Trends in Charpy-V Shelf Energy Degradation and Yield Strength Increase of Neutron-Embrittled Pressure Vessel Steels," NRL Report 7011, Dec. 1969; also, Nuclear Eng. and Design, 11(No. 3):427-446 (Apr. 1970).